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In-Vessel Retention Modeling Capabilities of SCDAP/RELAP5-3D©

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ABSTRACT

Molten core materials may relocate to the lower head of a reactor vessel in the latter stages of a severe accident. Under such circumstances, in-vessel retention (IVR) of the molten materials is a vital step in mitigating potential severe accident consequences. Whether IVR occurs depends on the interactions of a number of complex processes including heat transfer inside the accumulated molten pool, heat transfer from the molten pool to the reactor vessel (and to overlying fluids), and heat transfer from exterior vessel surfaces. SCDAP/RELAP5-3D© has been developed at the Idaho National Engineering and Environmental Laboratory to facilitate simulation of the processes affecting the potential for IVR, as well as processes involved in a wide variety of other reactor transients. In this paper, current capabilities of SCDAP/RELAP5-3D© relative to IVR modeling are described and results from typical applications are provided. In addition, anticipated developments to enhance IVR simulation with SCDAP/RELAP5-3D© are outlined.

NOMENCLATURE

$D_c(t)$	(structural) creep damage at time t (dimensionless)
L	characteristic length (m)
Pr	Prandtl number (dimensionless)
$\dot{Q}$	volumetric (decay) heat generation rate (W/m^3)
Ra	Rayleigh number for transient natural convection (dimensionless)
Ra'	Rayleigh number for steady state natural convection (dimensionless)
T	temperature (K)
a	position-dependent coefficient (W/m^2-^{o}C)
b	position-dependent coefficient (W/m^2-^{o}C^2)
c	position-dependent coefficient (W/m^2-^{o}C^3)
$f(T,\sigma)$	a function of temperature (T) and stress (σ) (dimensionless)
$f(\theta)$	a function of angle (θ) (dimensionless)
g	acceleration due to gravity (m/s^2)
$\bar{h}_{d,s}$	mean downward oxidic steady state convective heat transfer coefficient (W/m^2-^{o}C)
$\bar{h}_{d,t}$	mean downward oxidic transient convective heat transfer coefficient (W/m^2-^{o}C)
$h_{s,\mathrm{m}}$	sideward metallic convective heat transfer coefficient (W/m^2-^{o}C)
$h_{u,\mathrm{m}}$	upward metallic convective heat transfer coefficient (W/m^2-^{o}C)
$h_{u,s}$	upward oxidic steady state convective heat transfer coefficient (W/m^2-^{o}C)
$h_{u,t}$	upward oxidic transient convective heat transfer coefficient (W/m^2-^{o}C)
k	thermal conductivity (W/m-^{o}C)
q	heat flux (W/m^2)
q_{CHF}	critical heat flux (MW/m^2)
$t_r(t)$	time to creep rupture for a structure given the temperature and stress at time t (s)
α	thermal diffusivity (m^2/s)

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β coefficient of volumetric expansion (1/K)

ΔT temperature difference (^{o}C)

ΔT_w difference between the wall temperature and the saturation temperature for external flooding (^{o}C)

ΔT_{sub} degree of subcooling (^{o}C)

Δt current time step (s)

θ angle measured from the molten corium centerline (degrees)

ν kinematic viscosity (m^2/s)

σ (engineering) stress in a structure (ksi).

INTRODUCTION

In the initial phases of a severe reactor accident, some decrease or interruption in core cooling can lead to oxidation of core components, ballooning of fuel rod cladding, and ultimately, cladding failure. Core temperatures can significantly increase as a result of the exothermic oxidation reaction aggravated by flow reductions associated with cladding degradation. Low melting point core components (e.g., the control rods) can liquefy and then begin a downward relocation. Melting and downward relocation of the reactor fuel, with a corresponding release of fission products, can follow as core temperatures climb. Structural materials (e.g., the lower core support plate) may absorb energy sufficient to temporarily refreeze the relocating melt. However, without recovery of cooling capabilities, molten core materials will eventually accumulate in the lower head of the reactor vessel. A thermal attack of the lower head will follow.

At this point, a number of strategies have been proposed to prevent lower head failure and ensure in-vessel retention (IVR) of the molten corium. IVR of the corium is critical because an intact vessel provides a boundary limiting further spread of fission products. In addition, successful IVR prevents any potential challenge of containment integrity (from direct containment heating, steam explosions, and/or core/concrete interactions). IVR strategies include direct injection of water into the vessel (to enhance cooling off the corium surface, from cracks in the corium crust, and/or in narrow gaps that can form between the corium and the vessel), the use of core catchers (to effectively insulate the lower head from the corium and provide an engineered gap adjacent to the vessel), and various external reactor vessel cooling (ERVC) methods (primarily consisting of external sprays, external flooding, and/or external coatings to enhance heat transfer). These strategies (or combinations thereof) have been considered for application to existing, advanced, and Generation IV (GEN IV) reactors. Some possible configurations using these strategies are shown in Fig. 1.

The SCDAP/RELAP5-3D© computer code (The SCDAP/RELAP5-3D© Code Development Team, 2001) has been developed at the Idaho National Engineering and

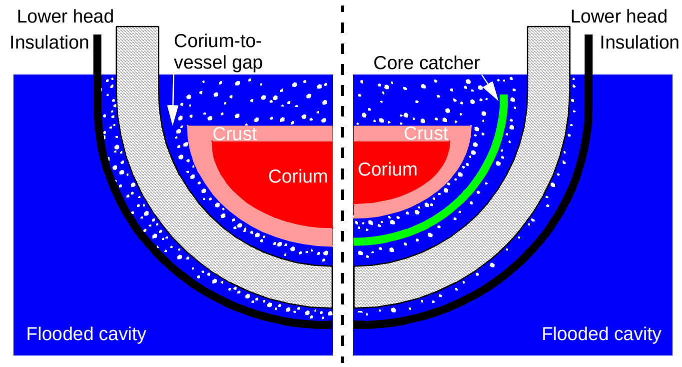

Figure 1. Configurations affecting IVR of corium.

Environmental Laboratory (INEEL) to facilitate simulation of processes affecting the potential for IVR, as well as processes involved in a wide variety of other reactor transients. Specifically, the code allows complete representation of reactors, reactor cooling systems, secondary (or balance-of-plant) systems, all related components (e.g., pumps, valves, turbines, condensers, pipes, tanks), and control systems during steady state, transient, and accident conditions. Heat transfer in the fuel and all other reactor structures can be modeled in two dimensions. Reactor kinetics and thermal-hydraulic flow models have full three dimensional capabilities. A wide variety of fluids can be analyzed throughout single- and two-phase conditions with or without the presence of non-condensable gases. Core heatup, oxidation of fuel rod cladding (with full thermal-hydraulic coupling of associated hydrogen production), clad ballooning and rupture, melting of core components, and relocation of molten core materials into the reactor vessel lower head can be calculated. The lower head thermal response and its potential failure can be predicted following relocation and accumulation of molten corium through use of a two dimensional finite element method. An advanced graphical user interface (GUI) is also available to assist with model development and analysis of code results. Typical GUI displays are shown in Fig. 2.

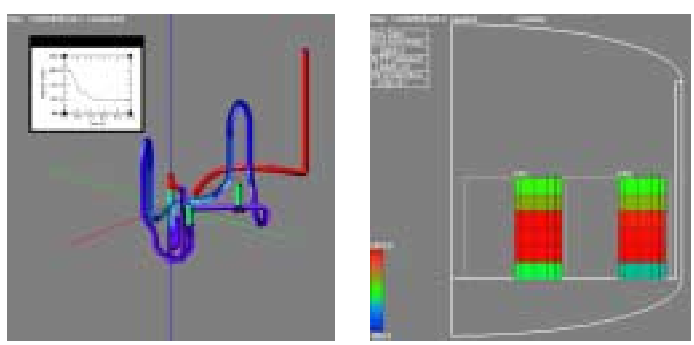

Figure 2. Typical GUI displays.

In this paper, current capabilities of SCDAP/RELAP5-3D© specifically related to molten corium IVR modeling are

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described and results from typical applications are provided. In addition, anticipated developments to enhance IVR simulation with SCDAP/RELAP5-3D© are outlined.

CURRENT CAPABILITIES

Current SCDAP/RELAP5-3D© IVR modeling capabilities begin with calculation of heat transfer in the corium pool as it accumulates in the lower head and progress through prediction of lower head failure.

Molten Pool Convective Heat Transfer

SCDAP/RELAP5-3D© contains two-dimensional finite element coding that is typically used to represent the lower head, including (interior) adjacent regions where corium can accumulate. Notable capabilities of this coding include calculation of (1) a corium bed that increases in depth sporadically with time, (2) spatially varying corium porosity, (3) phase changes for all materials in the finite element mesh, (4) conduction heat transfer through porous materials, (5) radiation heat transfer in porous materials, and (6) natural convection effects in the molten corium. Natural convection effects are of key interest here because this is a critical factor in lower head heating.

Relative to natural convection, two limiting molten corium configurations are possible. Those configurations include potentials for a stratified molten pool and a homogeneous (well mixed) molten pool. The stratified configuration could develop if relatively light metallic constituents separate from heavier oxidic constituents, leading to formation of a metallic layer on top of a supporting oxidic base. If viscous forces associated with natural convection exceed buoyancy forces, a more homogeneous pool could result. Because no criterion or correlation currently exists to determine whether molten corium materials stratify or remain well mixed during a given severe accident transient, SCDAP/RELAP5-3D© provides user flexibility to accommodate either configuration (which is particularly useful for completing sensitivity studies).

In the stratified configuration, the upward convective heat transfer coefficient in the metallic layer $(h_{u,\mathrm{m}})$ is based on experiments of Globe and Dropkin (1959) given by

$$h_{u,\mathrm{m}} = 0.051\frac{k}{L}Ra^{1/3} \tag{1}$$

where the depth of the metallic layer is used as the characteristic length (L) in the Rayleigh number (Ra) given by

$$Ra = \frac{g\beta L^3\Delta T}{\alpha\nu} \tag{2}$$

The user is allowed to select either a modified form of the Globe and Dropkin correlation (Theofanous, 1996) or the Churchill-Chu correlation (Churchill, 1975) for simulating convective heat transfer along the side of the metallic layer. Correlations for these coefficients $(h_{s,\mathrm{m}})$, which can "focus" heat transfer over the height of the metallic layer, are

$$h_{s,\mathrm{m}} = 0.15\frac{k}{L}Ra^{1/3}$$

$$h_{s,m} = \frac{k}{L}\left[0.825 + \frac{0.387Ra^{1/6}}{\{1 + (0.492/Pr)^{9/16}\}^{8/27}}\right]^2 \tag{3}$$

SCDAP/RELAP5-3D© tracks the location of all solid boundaries interfacing with molten corium in the lower head, which could include the vessel wall or positions of ever-changing crusts. Oxidic regions inside solid boundaries are subject to the effects of natural convection due to volumetric (decay) heating. As such, those regions are assumed to be thermally homogeneous, which is achieved through application of high conductivities.

Convective heat transfer from molten oxidic regions to solid boundaries is based on results from numerous experiments. For steady state natural convection, the coefficient for upward heat transfer $(h_{u,s})$ and the mean coefficient for downward heat transfer $(\bar{h}_{d,s})$ are given by

$$h_{u,s} = 0.345\frac{k}{L}Ra'^{\,0.233}$$

$$\bar{h}_{d,s} = 0.54f(\theta)\frac{k}{L}Ra'^{\,0.18} \tag{4}$$

based on correlations proposed by Steinberner and Reineke (1978) and Mayinger, et al. (1976), respectively. In this case, the modified Rayleigh number (Ra'), expressed as a function of the volumetric decay heat $(\dot{Q})$, is

$$Ra' = \frac{g\beta L^5\dot{Q}}{\alpha\nu k} \tag{5}$$

and the characteristic length (L) is taken to be the equivalent hemispherical radius of the molten region.

Experimental results of Jahn and Reineke (1983) and others indicate the local downward heat flux is a function of angular position (as indicated in Eq. 4). SCDAP/RELAP5-3D© accounts for that dependency consistent with the relationship shown in Fig. 3.

During transient natural convection in the molten corium pool, SCDAP/RELAP5-3D© includes appropriate modification of the convective heat transfer coefficients. Those modifications are reflected in

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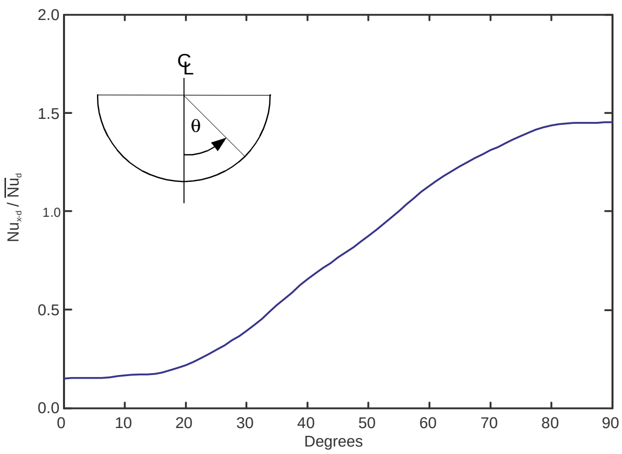

Figure 3. Nusselt number ratio as a function of angle (θ).

$$h_{u,t} = 0.25\frac{k}{L}Ra^{0.304}$$

$$\bar{h}_{d,t} = 0.472 f(\theta)\frac{k}{L}Ra^{0.22} \qquad (6)$$

Corium/Vessel Contact Resistance

Convective heat transfer from the molten corium pool directly to the wall of the lower head can occur. A more typical case, however, consists of convection to a corium crust (that solidified along the vessel wall) with conduction through the crust to a corium/lower head interface. SCDAP/RELAP5-3D© then provides a model to account for contact resistance heat transfer at the interface. The model is designed to simulate heat transfer associated with the imperfect contact of mating solid surfaces.

If corium at the interface is below its solidus temperature, an effective heat transfer coefficient is taken directly from user input (which is convenient for conducting sensitivity studies). If the corium is above its liquidus temperature, the coefficient is set to an arbitrarily high value. A value of 10,000 W/m^2-K is currently used, which is designed to simulate essentially perfect contact that would be expected if liquefied corium filled all voids associated with the surface roughness of the lower head.

Corium-to-Vessel Gap Cooling

Expansion and contraction of a reactor vessel and retained corium can occur following the relocation of molten core materials into the lower head. Generally, the lower head tends to expand and the corium tends to contract as a result of corium-to-vessel heat transfer, which can lead to the formation of gaps between the corium and the vessel. Gaps may also form when water initially trapped between the corium and the vessel expands during vaporization. Current experimental observations indicate the resulting gaps are quite narrow, typically in the range of 1 or 2 mm (Bang, et al., 1998 and Muriyama, 1998).

A simplified diagram of a typical corium-to-vessel gap configuration is shown in Fig. 4. A potential for development of countercurrent coolant flow (and overlying cross flow) exists as indicated. Such flows could be critical relative to limiting heatup and failure of the lower head.

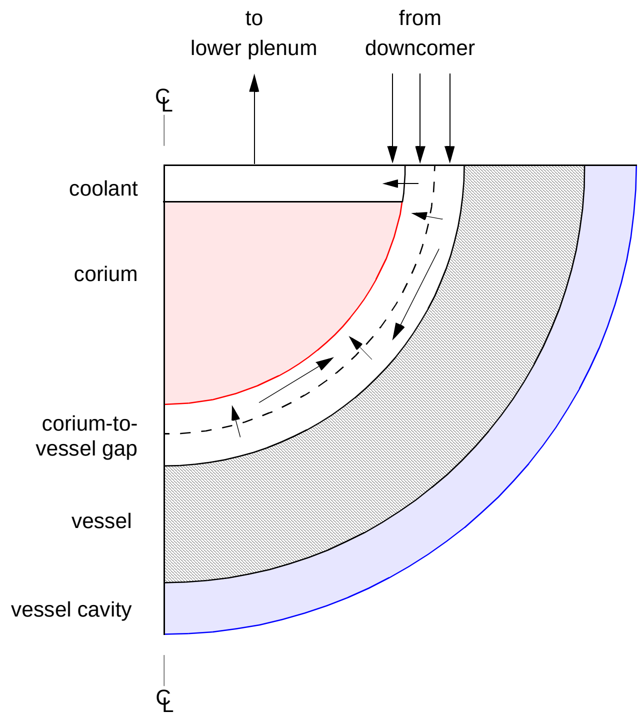

Figure 4. Corium-to-vessel gap configuration.

The current version of SCDAP/RELAP5-3D© contains models that allow simulation of a corium-to-vessel gap. The models offer a number of distinct features including capabilities to (1) specify any desired corium-to-vessel gap thickness, (2) specify gaps that vary in thickness in the axial direction (i.e., as a function of elevation), (3) easily add gaps to existing finite element meshes representing a reactor lower head, and (4) retain existing corium/vessel contact resistance heat transfer. The models automatically invoke full logic thermal-hydraulics with countercurrent flow limitations (CCFL), although this logic is currently based on correlations primarily developed for pipe flow.

ERVC

ERVC is the final aspect of heat transfer from the corium through the lower head to the surroundings. SCDAP/RELAP5-3D© currently provides two options for simulating ERVC including (1) user specification of external boundary conditions or (2) use of a model to simulate external flooding of the lower head with subcooled water. The user specification approach provides a convenient method for completing sensitivity studies. The external flooding model is based on experiments

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for subcooled nucleate boiling from the outer surfaces of an uninsulated submerged hemisphere in the Subscale Boundary Layer Boiling (SBLB) facility (Cheung, et al., 1997).

A set of subcooled nucleate boiling correlations are provided for calculation of heat flux (q) from a hemispherical surface in the form

$$q = a\Delta T_w + b\Delta T_w^2 + c\Delta T_w^3 \tag{7}$$

where a, b, and c are position-dependent coefficients. Equation (7) is valid when ΔT_w (the difference between the external vessel wall temperature and the bulk pool temperature) is between 4 ^oC and differential value associated with the critical heat flux (q_{CHF}) given by

$$q_{CHF} = 0.4(1 + 0.036\Delta T_{sub})(1 + 0.021\theta - (0.007\theta)^2) \tag{8}$$

In current versions of the code, Eq. (7) is used to calculate a position-dependent heat flux for ΔT_w = 4 ^oC. Linear interpolation between zero and the resulting flux (at ΔT_w = 4 ^oC) is performed to estimate the flux for any ΔT_w from 0 to 4 ^oC, which effectively simulates natural convection to the subcooled pool. Equations (7) and (8) are solved simultaneously to determine the position-dependent ΔT_w associated with the CHF. Equation (7) is then applied to determine the subcooled nucleate boiling flux for any ΔT_w between 4 ^oC and the ΔT_w at the CHF. The transition boiling heat flux is linearly extrapolated from the position-dependent CHF to a heat flux corresponding to a user defined heat transfer coefficient to vapor. The extrapolation is performed for any ΔT_w greater than ΔT_w at the CHF based on an estimate of the slope associated with transition regime experimental data.

Application of the approach is depicted in Fig. 5 as an example for several positions on the exterior surface of a lower head for a subcooling differential (ΔT_{sub}) of 10 ^oC and an assumed vapor coefficient of 375 W/m^2-K.

Lower Head Failure

A model based on creep rupture theory is used to calculate damage and the time to rupture for structural components specified by the user. As applied to the lower head, this represents the concluding step in determining whether IVR is successful. The model uses either Larson-Miller theory (Larson and Miller, 1952) or Manson-Haferd theory (Manson and Haferd, 1953), depending on the structural material and the calculated stress. In either case, damage (D_c) is given by

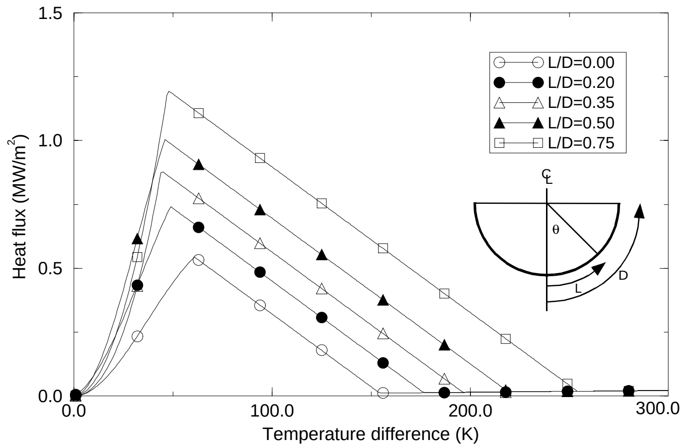

Figure 5. Heat flux to a subcooled pool from ERVC as a function of position and temperature difference.

$$D_c(t + \Delta t) = D_c(t) + \frac{\Delta t}{t_r(t)} \tag{9}$$

The time to rupture (t_r) for a structure (given the current temperature and stress) has the form

$$t_r = 10^{f(T, \sigma)} \tag{10}$$

where the function of temperature (T) and stress (σ) is determined by the applicable creep rupture theory.

Despite its simplicity, this model has been shown to reasonably correlate the onset of creep and the rupture time (Chu, et al., 1999).

APPLICATIONS AND INSIGHTS

Current versions of SCDAP/RELAP5-3D© have been applied to resolve numerous reactor safety issues. Selected applications and corresponding insights relative to IVR-specific issues are outlined below.

Corium-to-Vessel Gap Cooling Effects

SCDAP/RELAP5-3D© calculations have been performed with and without corium-to-vessel gaps to evaluate code behavior and the effects of heat transfer in the gaps. Normalized surface temperatures near the vessel centerline for a typical calculation (with an assumed uniform gap width of 2 mm) are provided in Figure 6.

Those results clearly indicate that vessel wall temperatures can be substantially reduced through the effects of gap cooling. Temperatures at the centerline location, like those at other vessel surface locations, tend to closely follow the coolant temperature. At the same time, significant vessel wall heating occurs without a gap (due to heat transfer through a user specified corium-to-vessel contact resistance).

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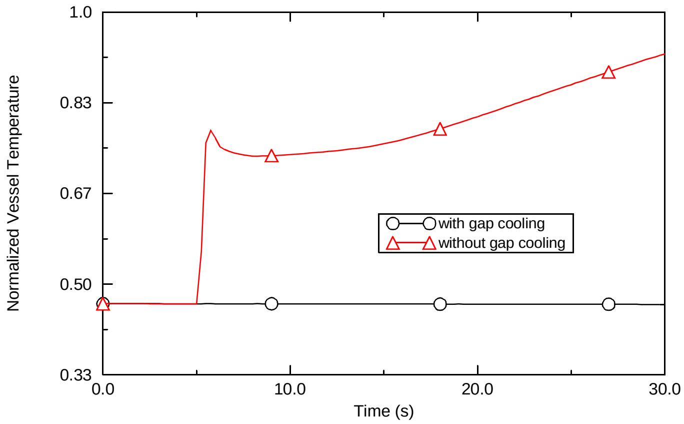

Figure 6. Temperatures of inner surfaces of a lower head with and without corium-to-vessel gap cooling.

Results shown in Figure 6 provide evidence that significant benefits relative to the IVR issue can be achieved by accounting for corium-to-vessel gap cooling. Although these insights are based on current heat transfer and CCFL correlations developed for pipes, similar trends are expected after narrow gap correlations have been implemented in the code. As discussed in later sections of this paper, the addition of narrow gap correlations is an area where code enhancement is anticipated.

ERVC Effects

SCDAP/RELAP5-3D© calculations have also been performed with and without external vessel flooding following the accumulation of molten corium in a reactor vessel lower head. Normalized results at the location of peak surface temperatures are shown in Figure 7.

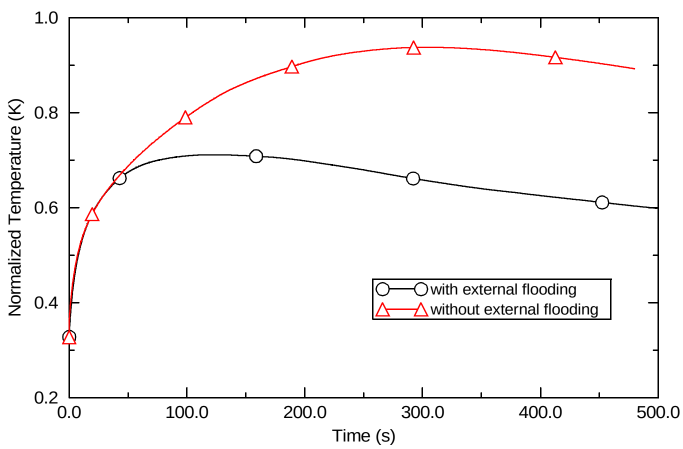

Figure 7. Vessel hot spots with and without external flooding.

The results provided in the figure were completed for vessels without penetrations or insulation. Furthermore, the power density of the relocated corium was assumed to be typical of existing light water reactors. However, the results indicate significant benefits may be achieved relative to the IVR issue if external flooding is used as an accident management strategy.

The addition of refined correlations for simulation of external flooding, and other ERVC schemes, is anticipated. At that time, additional SCDAP/RELAP5-3D© calculations will be completed to account for the effects of vessel insulation, penetrations, and the presence of coatings to enhance external vessel heat transfer.

ANTICIPATED DEVELOPMENTS

Enhancements to extend IVR modeling capabilities of SCDAP/RELAP5-3D© are currently anticipated in the three areas discussed below. Each of these enhancements are expected to significantly refine prediction of the lower head response and thereby improve prediction of the potential for IVR of accumulated corium.

Transient Corium-to-Vessel Gap Width

SCDAP/RELAP5-3D© currently includes provisions for modeling a corium-to-vessel gap with a fixed width based on user input (as previously indicated). An enhancement is contemplated to replace the fixed gap width with logic that will allow calculation of a variable gap width during each time step in a SCDAP/RELAP5-3D© simulation.

Calculation of the variable gap width will be based on the radial expansion/contraction of the reactor vessel lower head. Expansion/contraction of the lower head will be determined as functions of pressure- and temperature-induced stresses. Pressure- and temperature-induced stresses can be readily established using existing pressure and temperature results from the code. Lower head expansion/contraction can then be used to assign a gap width during each time step. The logic will be configured to determine this gap width as a function of position from the bottom to the top of the lower head. Accounting is also anticipated to cover lower head deformation through both elastic and plastic regions.

Full thermal-hydraulic feedback will be achieved through corresponding alteration of the coolant flow area during each time step. The SCDAP/RELAP5-3D© simulation will then be able to appropriately modify heat transfer and coolant flow as a function of the variable gap width, which will, in turn, affect lower head temperatures that are used to determine the width.

Narrow Gap Thermal-Hydraulic Correlations

Existing SCDAP/RELAP5-3D© models rely on heat transfer and fluid flow correlations primarily developed for pipes. The applicability of these correlations for simulating conditions in (very) narrow corium-to-vessel gaps has been recognized as a limitation. An enhancement of the code is anticipated as experimental data and more appropriate correlations emerge.

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The ultimate goal includes implementing a complete boiling curve and an appropriate countercurrent flow limitation (CCFL) model. The boiling curve is expected to resemble that shown in Fig. 8. A number of experimental and analytical investigations have been completed that may be used to characterize cooling in a narrow gap. (Asmolov, et al., 1999; Chang and Yao, 1983; Fujita, et al., 1988; Henry and Hammersley, 1996; Herbst, et al., 1999; Jeong, et al., 1998 and 1999; Kataoka, 1987; Koizumi, et al., 1997, 1999a, 1999b, and 2001; Monde, et al., 1982; Ohtake, 1998; Park, 1999; Schmidt, et al., 1998; Tanaka, et al., 2001) The Monde correlation was originally proposed to predict the maximum heat flux in narrow corium-to-vessel gaps (Suh, 1996a and Suh, 1996b).

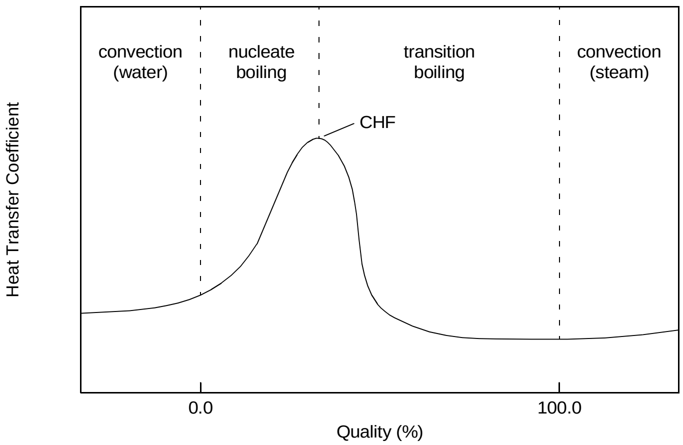

Figure 8. Expected characteristics of boiling heat transfer in a narrow channel.

Comparisons of currently available CHF correlations and data indicate significant variations in heat transfer predictions (Rempe, et al., 2001; Murase, et al., 2001). Some of these differences are due to the omission of CCFL effects, which clearly must be included in simulated narrow gap cooling. Other variations may be attributed to differences in test conditions (geometry, pressure, gap size, superheat, etc.), and correlations have been proposed that incorporate these effects.

Most of the current experiments have focussed on determining the CHF. Additional data for remaining transition points and correlations for each heat transfer regime will be developed based on available data, including new data that will be obtained from an on-going International Nuclear Energy Research Initiative (INERI) involving the INEEL, the Korea Atomic Energy Research Institute (KAERI), as well as universities in the United States and Korea (INEEL, 2001).

ERVC Correlations

The SBLB tests used to develop the correlations shown in Fig. 5 used a 0.305 m diameter hemispherical test vessel, approximately 1/10th the scale of most light water reactor (LWR) vessels. Several larger scale, two-dimensional facilities, such as the University of California at Santa Barbara (UCSB) ULPU facility (Theofanous, et al., 1996) and the Commissariat A L'Energie Atomique (CEA) SULTAN facility (Rougé, 1997 and Rougé, et al., 1999) have obtained ERVC relationships as function of angle. In Cheung, et al., (1997), a scaling law is proposed that allows one to extrapolate the SBLB data to other vessel scales, coolant subcooling, coolant velocities, gravity heads, and coolant pressures. This scaling law is compared to relationships developed from other larger scale test facilities in Fig. 9. Considering the large variation in approaches used to obtain these relationships, correlation predictions are fairly close (less than 20% difference for angles greater than 75º and H/R=3).

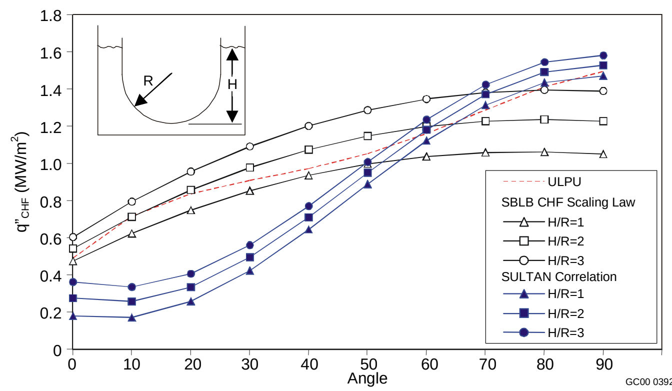

Figure 9. Comparison of correlation predictions from SULTAN, ULPU, and SBLB tests.

In later SBLB tests, reactor vessel insulation effects were investigated (Cheung, et al., 1998, and Cheung and Liu, 1999). SBLB tests were conducted to investigate two different insulation geometries - one to represent the insulation for the Westinghouse AP600 and one to represent proposed insulation/ penetration configurations for the Korean Advanced Power Reactor 1400 MW_e (APR1400). Results from SBLB tests simulating the impact of insulation with results from earlier uninsulated tests are compared in Fig. 10. As shown, inclusion of the insulation structure led to higher CHF values at all locations. However, at a location near where a minimum gap existed between the insulation and the hemispherical test vessel (at approximately 45º), a CHF value that was lower than the CHF at the bottom center of the vessel was measured. At this location, the vapor slug tended to occupy the entire cross-sectional flow area, preventing the supply of water to the heating surface. This resulted in premature dryout of the surface, leading to a considerably smaller value of the local CHF. Hence, this suggests that minimum gap regions may have the potential to lead to hot spots on the vessel wall during a severe accident.

Based on the above information, the following new options will be implemented into SCDAP/RELAP5-3D© to improve

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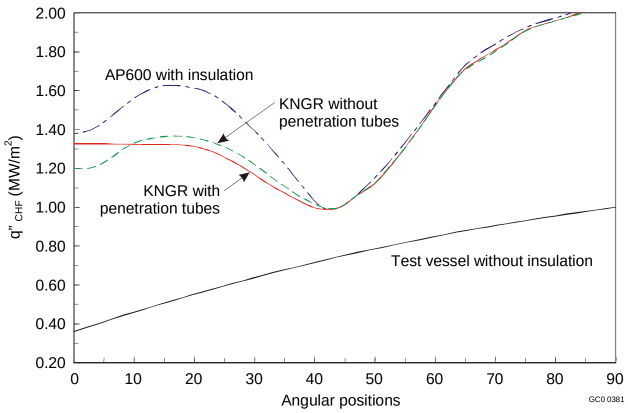

Figure 10. Variation of CHF measured in various SBLB geometries.

ERVC modeling capabilities. First, the scaling relationship proposed by Cheung, et al., (1997) will be implemented to replace the existing correlations. For uninsulated vessels, this scaling law will allow ERVC to be predicted for a range of conditions. Furthermore, additional options will be implemented into SCDAP/RELAP5-3D© that allows selected reactor insulation configurations and vessel coatings to be modeled.

SUMMARY

Molten core materials may relocate to the lower head of a reactor vessel in the latter stages of a severe accident. Under such circumstances, IVR of the molten materials is a vital step in mitigating potential severe accident consequences. SCDAP/RELAP5-3D© has been developed at the INEEL to facilitate simulation of the processes affecting the potential for IVR, as well as processes involved in a wide variety of other reactor transients. Current capabilities of SCDAP/RELAP5-3D© relative to IVR modeling include simulation of molten pool convective heat transfer, corium/vessel contact resistance, corium-to-vessel gap cooling, ERVC, and lower head failure. SCDAP/RELAP5-3D© results using current capabilities indicate significant benefits relative to the IVR issue may be derived from accounting for corium-to-vessel gap cooling and application of external reactor vessel flooding as an accident management strategy. Similar trends are expected to hold following anticipated code enhancements to allow calculation of a transient-dependent corium-to-vessel gap width and to incorporate narrow gap thermal-hydraulic correlations and refined correlations for ERVC.

ACKNOWLEDGMENTS

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